Nuclear Engineering and Design 241 (2011) 1137–1144

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

Sensitivity analysis of the MASLWR helical coil steam generator using TRACE

F. Mascari[a,*], G. Vella[a], B.G. Woods[b], K. Welter[c], J. Pottorf[c], E. Young[c], M. Adorni[d], F. D'auria[d]

[a] Dipartimento di Ingegneria Nucleare, Università degli Studi di Palermo, Viale delle Scienze, Edificio 6, 90128 Palermo, Italy
[b] Department of Nuclear Engineering and Radiation Health Physics, Oregon State University, 116 Radiation Center, Corvallis, OR 97331-5902, USA
[c] NuScale Power Inc., 201 NW Third Street, Corvallis, OR 97330, USA
[d] San Piero a Grado – Nuclear Research Group (SPGNRG), University of Pisa, Italy

ARTICLE INFO

Article history:
Received 1 February 2010
Received in revised form 3 May 2010
Accepted 4 May 2010
Available online 15 June 2010

ABSTRACT

Accurate simulation of transient system behavior of a nuclear power plant is the goal of systems code calculations, and the evaluation of a code's calculation accuracy is accomplished by assessment and validation against appropriate system data. These system data may be developed either from a running system prototype or from a scaled model test facility, and characterize the thermal hydraulic phenomena during both steady state and transient conditions. The identification and characterization of the relevant thermal hydraulic phenomena, and the assessment and validation of thermal hydraulic systems codes, has been the objective of multiple international research programs. The validation and assessment of the best estimate thermal hydraulic system code TRACE against the Multi-Application Small Light-Water Reactor (MASLWR) Natural Circulation (NC), helical coil Steam Generator (SG), Nuclear Steam Supply System (NSSS) design is a novel effort, and is the topic of the present paper. Specifically, the current work relates to the assessment and validation process of TRACE code against the NC database developed in the OSU-MASLWR test facility. This facility was constructed at Oregon State University under a U.S. Department of Energy grant in order to examine the NC phenomena of importance to the MASLWR reactor design, which includes an integrated helical coil SG. Test series have been conducted at this facility in order to assess the behavior of the MASLWR concept in both normal and transient operation and to assess the passive safety systems under transient conditions. In particular the test OSU-MASLWR-002 investigated the primary system flow rates and secondary side steam superheat, used to control the facility, for a variety of core power levels and Feed Water (FW) flow rates. This paper illustrates a preliminary analysis, performed by TRACE code, aiming at the evaluation of the code capability in predicting NC phenomena and heat exchange from primary to secondary side by helical SG in superheated condition and to evaluate the fidelity of various methods to model the OSU-MASLWR SG in TRACE. The analyses of the calculated data show that the phenomena of interest of the OSU-MASLWR-002 test are predicted by the code and that one of the reasons of the instability of the superheat condition of the fluid at the outlet of the SG is the equivalent SG model used to simulate the different group of helical coils. The SNAP animation model capability is used to show a direct visualization of selected calculated data.

1. Introduction

Accurate simulation of transient system behavior of a nuclear power plant is the goal of systems code calculations, and the evaluation of a code's calculation accuracy is accomplished by assessment and validation against appropriate system data. These system data may be developed either from a running system prototype or from a scaled model test facility, and characterize the thermal hydraulic phenomena during both steady state and transient conditions. The identification and characterization of the relevant thermal hydraulic phenomena, and the assessment and validation of thermal hydraulic systems codes, has been the objective of multiple international research programs.

In this framework Oregon State University (OSU) has constructed, under a U.S. Department of Energy grant, a system-level

Abbreviations: ADS, Automatic Depressurization System; CHF, Critical Heat Flux; CL, Cold Leg; FW, Feed Water; HL, Hot Leg; HPC, High Pressure Containment; IAEA, International Atomic Energy Agency; ICSP, International Collaborative Standard Problem; LOCA, Loss of Coolant Accident; LP, Lower Plenum; LWR, Light-Water Reactor; MASLWR, Multi-Application Small Light-Water Reactor; MS, Main Steam; NC, Natural Circulation; NPP, Nuclear Power Plant; NSSS, Nuclear Steam Supply System; OECD, Organization for Economic Cooperation and Development; OSU, Oregon State University; PARCS, Purdue Advanced Reactor Core Simulator; PKL, Primärkreisläufe (Test Facility); PRZ, Pressurizer; PWR, Pressurized Water Reactor; RHRS, Residual Heat Removal System; RELAP, Reactor Excursion and Leak Analysis Program; ROSA/LSTF, ROSA Large Scale Test Facility; RPV, Reactor Pressure Vessel; SESAR, Senior Group of Experts on Nuclear Safety Research; SETH, SESAR Thermal Hydraulics; SBLOCA, Small Break Loss of Coolant Accident; SG, Steam Generator; SNAP, Symbolic Nuclear Analysis Package; TRACE, TRAC/RELAP Advanced Computational Engine; UP, Upper Plenum; USNRC, U.S. Nuclear Regulatory Commission.

* Corresponding author. Tel.: +39 3881135591; fax: +39 091232215.
E-mail addresses: mfulvio78@yahoo.it, fmascari@din.unipa.it (F. Mascari).

0029-5493/$ – see front matter
doi:10.1016/j.nucengdes.2010.05.002

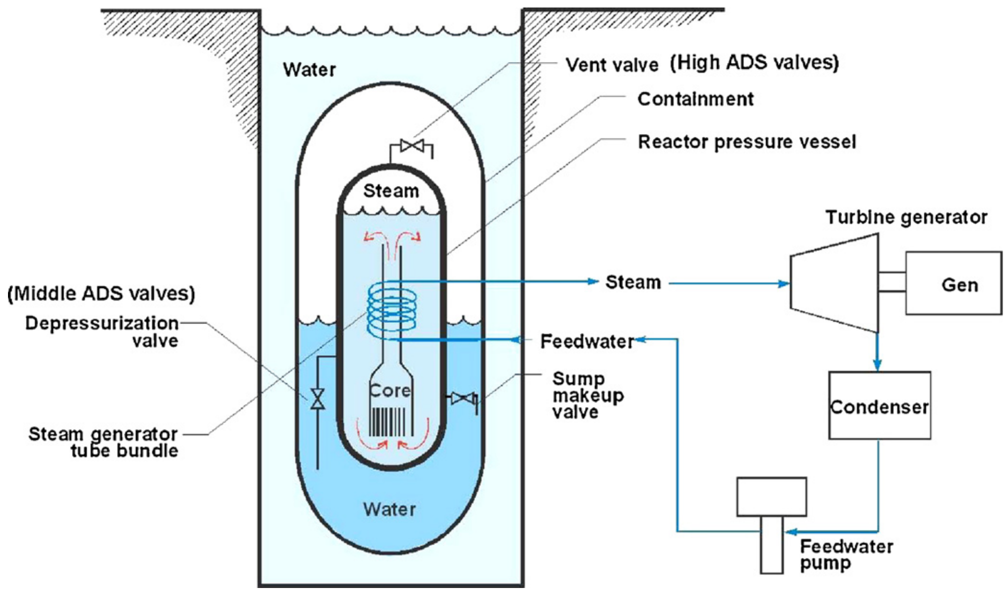

Fig. 1. MASLWR conceptual design layout (Modro et al., 2003; Reyes et al., 2007).

test facility to examine NC phenomena of importance to the MASLWR reactor design, developed by Idaho National Engineering and Environmental Laboratory, OSU and Nexant–Bechtel.

The MASLWR (Modro et al., 2003), Fig. 1, is a small modular pressurized light-water reactor relying on NC during both steady state and transient operation, which includes an integrated SG consisting of banks of vertical helical tubes contained within the Reactor Pressure Vessel (RPV) and located in the upper region of the vessel outside of the Hot Leg (HL) chimney. The primary coolant flows outside the SG tubes, and the FW is fully vaporized resulting in superheated steam at the SG exit. MASLWR's safety systems are designed to operate passively. The RPV is surrounded by a cylindrical containment, partially filled with water, which provides pressure suppression and liquid makeup capabilities. The RPV can be depressurized using the Automatic Depressurization System (ADS), which consists of six valves discharging into various locations within the containment. The entire containment vessel is submerged in a pool of water that acts as the ultimate heat sink. The MASLWR has a net output of 35 MWe. Its small size makes the prototypical MASLWR relatively portable and thus well suited for employment in smaller electricity grids. These smaller electricity grids may be found in developing or remote regions.

The planned work related to the OSU-MASLWR test facility will be not only to specifically investigate the MASLWR concept design further but also advance the broad understanding of integral NC reactor plants and accompanying passive safety features as well. Furthermore an IAEA International Collaborative Standard Problem (ICSP) (Woods and Mascari, 2009) on the "Integral PWR Design Natural Circulation Flow Stability and Thermo-Hydraulic Coupling of Containment and Primary System during Accidents" will be executed in the facility. The purpose of this IAEA ICSP is to provide experimental data on single/two-phase flow instability phenomena under NC conditions and coupled containment/reactor vessel behavior in integral-type reactors. This data can be used to assess thermal hydraulic codes for reactor system design and analysis.

In order to analyze the thermal hydraulic behavior of LWR reactors, the USNRC has maintained four codes: RAMONA, RELAP5, TRAC-B and the TRAC-P (Boyack and Ward, 2000). In the last years the NRC has developed an advanced best estimate thermal hydraulic system code, called TRAC/RELAP Advanced Computational Engine or TRACE (Cheng et al., 2009; Reyes, 2005; TRACE V5.0, 2008), to perform best estimate analysis for LWR. Different studies using the TRACE code have been developed in the recent years. A TRACE model of Almaraz NPP has been used to study a loss of RHRS at midloop operation (Queral et al., 2008). A TRACE model of the Maanshan PWR NPP has been used to evaluate its effectiveness by simulating a turbine trip and load reduction transients and comparing the results with Maanshan NPP data (Wang et al., 2009). A TRACE model of the PKL test facility has been used to simulate the PKL III E3.1 test (Jasiulevicius, 2005) – loss of RHRS in midloop operation with the reactor coolant system closed – in the framework of OECD SETH/PKL benchmark on test E3.1 (Bucalossi, 2006). A TRACE model of ROSA/LSTF test facility has also recently been used to simulate a RPV upper head SBLOCA test (Freixa and Manera, 2010). Furthermore, the analysis of a test of "inadvertent actuation of 1 submerged ADS valve" (OSU-MASLWR-001 test), performed in the OSU-MASLWR test facility, has been developed using TRACE, RELAP5/Mod3.3, and RELAP5-3D code (Pottorf et al., 2009).

In the framework of the performance assessment and validation of thermal hydraulic codes, this paper illustrates a preliminary analysis, performed by TRACE code (TRACE V5.0 Patch 01), aiming at the evaluation of the code capability in predicting NC phenomena and heat exchange from primary to secondary side by helical SG in superheated condition and to evaluate the fidelity of various methods to model the SG in TRACE by simulating the OSU-MASLWR-002 test.

2. Description of the OSU-MASLWR facility

2.1. OSU-MASLWR test facility overview

The OSU-MASLWR test facility (Galvin, 2007; Reyes et al., 2007), shown in Fig. 2, is scaled at 1:3 length scale and 1:254 volume scale, is constructed entirely of stainless steel, and it is designed for full pressure (11.4 MPa) and full temperature (590 K) prototype operation. The facility includes the primary and secondary circuit and the containment structure. The primary circuit includes the RPV and the ADS blowdown lines, vent lines and sump recirculation lines. The internal components of the RPV, Fig. 3, are the core, the HL riser, the Upper Plenum (UP), the Pressurizer (PRZ), the SG primary side, the Cold Leg (CL) downcomer and the Lower Plenum (LP). The secondary circuit includes the FW treatment and storage system, the main FW pump, the main FW system supply lines, the SG sec-

Fig. 2. Photograph of OSU-MASLWR test facility (Galvin, 2007).

ondary side internal to the vessel, the Main Steam (MS) system and associated FW and steam valves.

Two vessels, a High Pressure Containment (HPC) vessel and a cooling pool vessel with an heat transfer surface between them to establish the proper heat transfer area, are used to model the containment structure, in which the RPV sits, as well as the cavity within which the containment structure is located. In addition to the physical structures that comprise the test facility, there are data acquisition, instrumentation and control system. Auxiliary lines and systems are present in the facility.

2.2. OSU-MASLWR RPV overview

The primary flow, Fig. 3, exits the un-rodded LP region, below the downcomer, radially inward into the rodded but unheated LP region, then upward into bottom of the core via a lower core flow plate. After leaving the core, the flow enters the chimney of the HL riser that creates a riser/downcomer configuration to enable NC. After leaving the top of the HL riser, the flow enters the UP that directs the flow radially outward and then down into the SG coil bundle of the SG section. After leaving the SG section, the flow continues downward into the CL downcomer region. This is an annular region bounded by the RPV wall on the outside and the HL riser on the inside, and the flow area reduces at the HL riser cone. The flow exits the CL downcomer region into the LP to complete the primary flow circuit.

The core is modelled with 56 cylindrical heater rods distributed in a 1.86 cm pitch square array with a 1.33 pitch to diameter ratio. The nominal power of each heater rod is 7.1 kW resulting in a maximum core power of 398 kW. The core is shrouded to ensure that all flow enters the core via the bottom and travels the entire heated length.

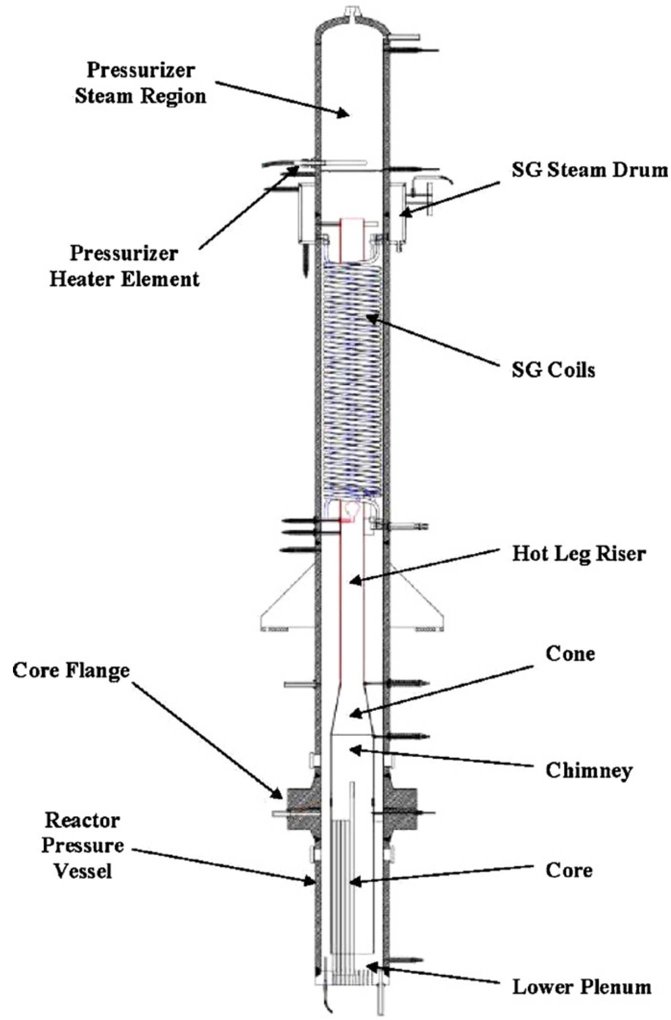

Fig. 3. OSU-MASLWR RPV key areas (Galvin, 2007; Reyes et al., 2007).

The UP is separated from the heated upper PRZ section by a thick baffle plate having eight holes, spaced uniformly around the baffle plate periphery, which allow free communication of the PRZ to the remainder of the RPV during normal operation and for volume surges into and/or out of the PRZ due to transients. In the PRZ there are three heater elements, each 4 kW, that are modulated by the test facility control system to maintain nominal primary system pressure at the desired value (nominally 11.4 MPa).

2.3. OSU-MASLWR SG overview

The SG of the facility is a once through heat exchanger and is located within the RPV in the annular space between the HL riser and the inside surface of the RPV. The tube bundle, Fig. 4, is a helical coil consisting of 14 tubes with a total heated length of 86 m. There are three separate parallel coils of stainless steel tubes. The outer and middle coils consist of five tubes each while the inner coil consists of four tubes. Each coil is joined at a common inlet header and each of them exhausts the superheated steam into a common steam drum from where it is subsequently exhausted to atmosphere via the MS system.

The FW, pumped in the SG from an FW storage tank by a positive displacement pump, enters at the bottom of the SG and boils off after travelling a certain length in the SG. This boil off length is a function of both core power and MFW flow rate. Nominally, the boil off length is approximately 40% shorter than the actual length

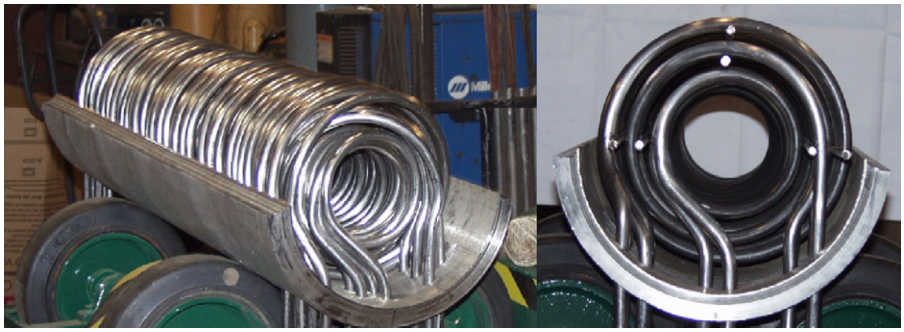

Fig. 4. Photographs of SG coil bundle arrangement (Galvin, 2007).

Table 1
SG geometrical data (Galvin, 2007).

	Inner	Mid	Outer
Number of tubes in bank	4	5	5
Number of rotations from feed inlet to steam outlet	13	9.5	7.5
Average tube length of the bank (m)	6.05	6.15	6.21
Total average tube length (m)	6.21	6.30	6.36
Total tube bank surface area (m^2)	1.209	1.532	1.550
Individual tube outside diameter (m)	0.0159	0.0159	0.0159
Individual tube wall thickness (m)	0.00165	0.00165	0.00165

of the SG tubes so the steam will leave the SG superheated. The value of the degree of the steam superheat is changed in order to control the facility. Table 1 shows some principal SG geometrical data.

2.4. OSU-MASLWR testing campaign

The first tests conducted on the OSU-MASLWR facility (Modro et al., 2003; Reyes et al., 2007) were in support of the MASLWR concept design verification. Table 2 shows, in chronological order, the type of test already conducted in the facility.

The purpose of the test OSU-MASLWR-001, a design basis accident for MASLWR concept design, was to determine the behavior of the RPV and containment pressure following an inadvertent actuation of 1 middle ADS valve. The test successfully demonstrated the behavior of the MASLWR during one of its design basis accident. The normal open sequence used in the MASLWR for the ADS valves is the submerged lines first, after the high containment lines and finally the sump recirculation lines. This sequence minimizes the rise in containment pressure since a large fraction of the energy transferred to the containment is direct into the subcooled containment coolant. However, if the high containment lines are actuated first, the raise in containment pressure will be larger than the previous one and, anyway, the chocked flow in the high containment line will limit the rate at which the containment and vessel pressure equalize. Therefore the purpose of the test OSU-MASLWR-003B is to acquire the pressure transient in the containment and the primary system during the inadvertent actuation of the high containment ADS vent line. This test represents a beyond design basis accident scenario for the MASLWR and demonstrates the vessel containment coupling and containment vessel condensation behavior during the transient.

Table 2
Summary of the previous OSU-MASLWR testing program.

Name of the test	Type of test
OSU-MASLWR-001	Inadvertent actuation of 1 submerged ADS valve
OSU-MASLWR-002	Natural circulation at core power up to 210 kW
OSU-MASLWR-003A	Natural circulation at core power of 210 kW (continuation of test 002)
OSU-MASLWR-003B	Inadvertent actuation of 1 high containment ADS valve

The test OSU-MASLWR-002 and OSU-MASLWR-003A investigated the primary system flow rates and secondary side steam superheat for a variety of core power levels and FW flow rate. OSU-MASLWR-002 stepped power level incrementally up to 165 kW, varying FW flow rate at each power level, and OSU-MASLWR-003A was an extended 210 kW steady test establishing initial conditions for the following test OSU-MASLWR-003B. During these two tests seven different core powers as well as nine different FW flow rates were used.

As the slope of the MS superheat curve increases if the value of the core power increases and decreases if the value of the FW flow rate increases, the target of these tests was to acquire primary system flow rate and secondary side steam superheat for different core power and FW flow rate. The difference between the MS saturation temperature and the measured MS temperature is used to estimate the value of the MS superheat. The phenomena (NEA\CSNI\R(96)17, 1996) of interest in these tests are the single phase NC phenomena, the heat transfer in covered core, the heat transfer in SG primary and secondary side and the superheating in secondary side for a variety of primary and secondary operation conditions.

As the test OSU-MASLWR-002 gives the wider number of information about these phenomena, this is the test chosen for evaluating the TRACE code capability in predicting NC phenomena and heat exchange from primary to secondary side by helical SG in superheated condition and for evaluating the fidelity of various methods to model the OSU-MASLWR SG in TRACE.

3. Code application

3.1. TRACE code

In order to analyze the thermal hydraulic behavior of LWR reactors, the USNRC has maintained four codes, the RAMONA, the RELAP5, the TRAC-B and the TRAC-P (Boyack and Ward, 2000). In the last years the NRC is developing an advanced best estimate thermal hydraulic system code, by merging, among other things, the capability of the previous codes into a single code. This new code is called TRAC/RELAP Advanced Computational Engine or TRACE (Cheng et al., 2009; Reyes, 2005; TRACE V5.0, 2008), and is a component-oriented code designed to perform best estimate analysis for LWR. In particular this code is developed to simulate operational transient, LOCA, other transient typical of the LWR and to model the thermal hydraulic phenomena taking place in the experimental facilities used to study the steady state and transient behavior of reactor systems.

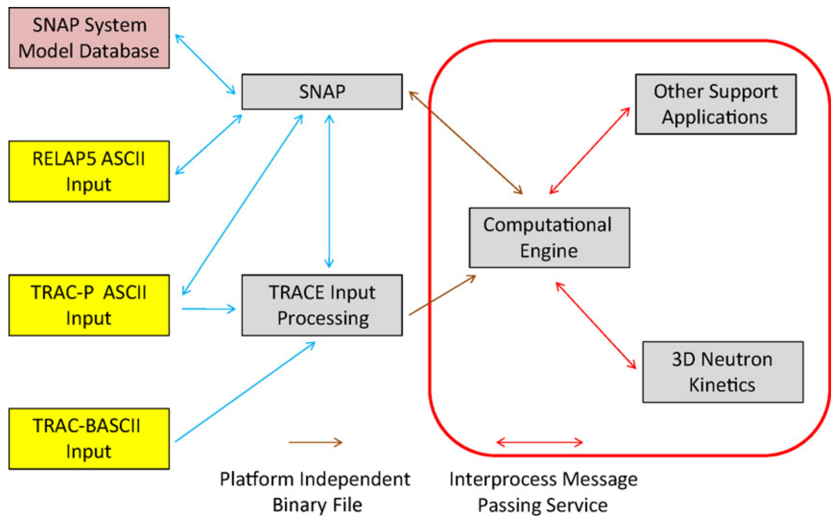

Fig. 5. TRACE/SNAP environment architecture (Staudenmeier, 2004).

TRACE is a finite volume, two fluid, code with 3D capability which gives user the possibility to model heat structures and control systems that interact with the component models. It can be run coupled with the 3D reactor kinetics code PARCS as well. TRACE can be used together with a user-friendly front end, Symbolic Nuclear Analysis Package (SNAP, 2007), able to support the user in the development and visualization of the model, to show a direct visualization of selected calculated data using the animation model capability, and accepts existing RELAP5 and TRAC-P input. The computational complexity of a generic TRACE model is only limited by the availability of the computer memory.

The code is based on two fluid, two-phase field equations. This set of equations consists of the conservation laws of mass, momentum and energy for liquid and gas fields. The resulting equation set is coupled to additional equations for non-condensable gas, dissolved boron, control systems and reactor power. Relations for wall drag, interfacial drag, wall heat transfer, interfacial heat transfer, equation of state and static flow regime maps are used for the closure of the field equations. The interaction between the steam-liquid phases and the heat flow from solid structures is also considered. These interactions are in general dependent on flow topology and for this purpose a special flow regime dependent constitutive-equation package has been incorporated into the code. TRACE uses a pre-CHF flow regime, a stratified flow regime and a post-CHF flow regime. In order to study the thermal history of the structures the heat conduction equation is applied to different geometry. A 2D (r and z) treatment of conduction heat transfer is taken into account as well.

A finite volume numerical method is used to solve the partial differential equations governing the two-phase flow and heat transfer. By default, a multi-step time-differencing procedure that allows the material Courant-limit condition to be exceeded is used to solve the fluid-dynamics equations. Fig. 5 shows the TRACE/SNAP environment architecture.

3.2. OSU-MASLWR TRACE model

The OSU-MASLWR TRACE model (Mascari et al., 2008, 2009; Pottorf et al., 2009), made using SNAP and shown in Fig. 6, is developed in order to assess and validate the TRACE code against the NC database developed in the OSU-MASLWR test facility. This model will be used in the envisaged ICSP.

The nodalization models the primary and the secondary circuit, the HPC, the heat transfer plate and the cooling pool vessel. The ADS blowdown lines, vent lines and sump recirculation lines are modelled as well. The "slice nodalization" technique is adopted in order to improve the capability of the code to reproduce NC phenomena. This technique consists in realizing the mesh cells of different nodalization zones, at the same elevation, with the same cell length.

The primary circuit of the TRACE model comprises the core, the HL riser, the UP, the PRZ, the SG primary side, the CL downcomer and the LP. After leaving the top of the HL riser, the flow enters the UP divided into two thermal hydraulic regions connected to the PRZ. After living the UP the flow continues downward through the SG primary section and into the CL downcomer region. The core is modelled with one thermal hydraulic region thermally coupled with one equivalent active heat structure simulating the 56 electric heaters. The PRZ is modelled with two hydraulic regions, connected by different single junctions, in order to allow potential NC/convection phenomena. The three different PRZ heater elements are modelled with one equivalent active heat structure. The thick baffle plate is modelled as well. The direct heat exchange by the internal shell between the hotter fluid in the ascending riser and the colder fluid in the descending annular downcomer is modelled by heat structures thermally coupled with these two different hydraulic regions. The RPV, HPC and cooling pool vessel shell and the connected insulation are modelled.

In the first model, considered in this analysis (REF), Fig. 6, the SG coils are modelled with three different "equivalent" oblique group of pipes in order to simulate the three separate parallel coils of tubes. In the second model (SEN1) the SG coils are simulated with three different "equivalent" vertical group of pipes. In these two models each equivalent group of pipe is thermally coupled, by a heat structure, with the SG primary side section. In the third model (SEN2) the SG coils are simulated with only one "equivalent" vertical group of pipes thermally coupled, by one equivalent heat structure, with the SG primary side section.

3.3. TRACE model qualification process

A nodalization, representing an actual system (integral test facility or nuclear power plant), can be considered qualified when: (a) it has a geometrical fidelity with the involved system, (b) it reproduces the measured nominal steady state conditions of the system, and (c) it shows a satisfactory behavior in time-dependent conditions. Taking into account these statements, a standard procedure (Bonuccelli et al., 1993) for the nodalization qualification, has been considered.

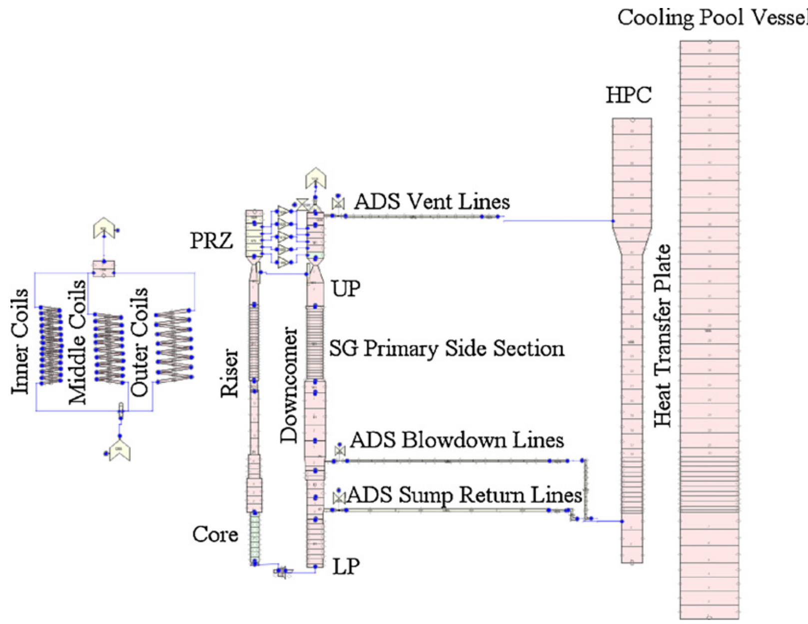

Fig. 6. OSU-MASLWR TRACE model (REF).

The OSU-MASLWR nodalizations qualification process is still in progress, because the facility experimental characterization will be conducted in the framework of the envisaged ICSP. In particular several important facility operational characteristics, like pressure drop along the primary loop and local and integral heat losses, determined to be of importance during the planned ICSP experiments, will be evaluated. The nodalization models, here presented, are still preliminary because some geometrical data and the complete instrument characterization and location will be delivered in the ICSP framework as well. Therefore the current results are preliminary and should not be used for the code assessment, but are able to show the TRACE capability to reproduce NC phenomena and heat exchange from primary to secondary side by helical SG in superheated condition.

3.4. OSU-MASLWR-002 reference and sensitivity calculation results

The analyses of the calculated data shows that the phenomena of interest in the OSU-MASLWR-002 test are predicted by the code. Therefore the primary system flow rates and the secondary side steam superheat, for a variety of core power levels and FW flow rates, are collected by the TRACE analysis and compared with the experimental data.

The heat transfer in covered core and the single phase NC phenomena are qualitatively predicted by the code as it is shown by the inlet/outlet fluid core temperature, the primary volumetric flow rate and the difference between the core inlet and outlet fluid temperature (delta *T* core). The inlet/outlet fluid core temperature, Fig. 7, show a qualitative agreement but a general overestimation compared with the experimental data. Therefore, in the TRACE simulation, the primary circuit stores more energy compared with the experimental data.

The primary NC volumetric flow rate behavior, Fig. 8, is qualitatively predicted by the code but it shows a general underestimation compared to the experimental data in particular the last 500 s of the transient. One of the main reasons is the primary side pressure drop distribution calculated by TRACE model. In order to qualify these pressure drops, it is necessary to develop experimental tests aiming at the evaluation of the distribution of the pressure drop along the primary loop for different steady state condition characterized by different primary volumetric flow rate. This activity will be performed in the framework of the envisaged ICSP.

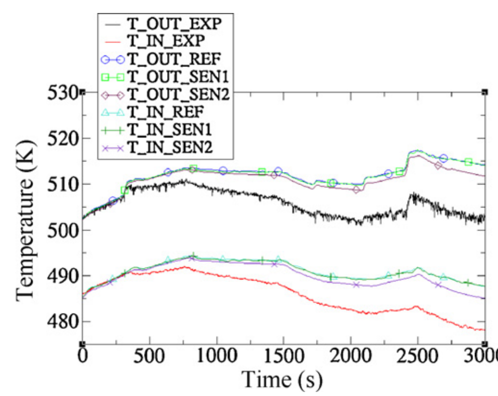

Fig. 7. Experimental data versus code calculations for fluid temperature at the core outlet/inlet.

The core delta *T*, related to the primary volumetric flow rate and power and thermo physical condition of the primary fluid, is qualitatively predicted by the code but it shows a general overestimation compared to the experimental data in particular the last 500 s of the transient, Fig. 9. This is in a physical agreement with the primary volumetric flow rate behavior.

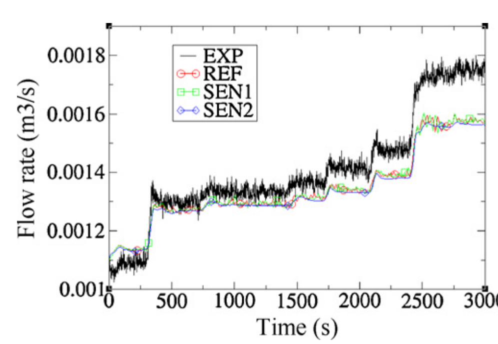

Fig. 8. Experimental data versus code calculations for primary volumetric flow rate.

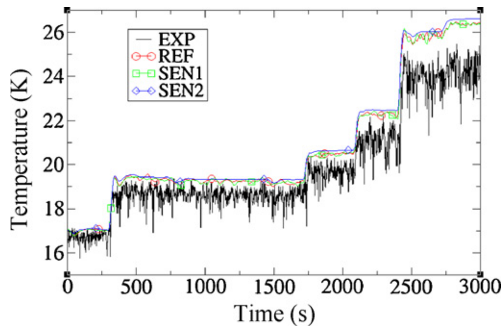

Fig. 9. Experimental data versus code calculations for core delta T.

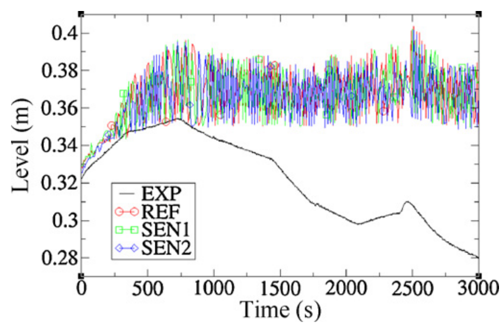

Fig. 10. Experimental data versus code calculations for PRZ level.

The time dependence of the core power is imposed like boundary condition. The PRZ heaters are used, during the calculation, to control the PRZ pressure. The PRZ level is qualitatively predicted by the code but shows a continuous small oscillation during all the transient, Fig. 10. The PRZ pressure behavior shows discrepancies compared with the experimental data, Fig. 11. More investigations need the PRZ nodalization model (hydraulic volume and heat structure) and the logic to control the PRZ pressure using the PRZ heaters.

By analyzing the experimental data, related to the flow temperature after the SG coils primary side section and the core inlet temperature, it is evident that the direct heat exchange, through the internal shell, between the fluid ascending the HL and the fluid descending the CL, is a crucial parameter for the evaluation of the core inlet temperature and therefore the core outlet temperature. In fact, the experimental data show that, along the downcomer region, the fluid increases its temperature between the end of the SG primary side section and the core inlet. An overestimation or an underestimation of this phenomenon creates an increase or decrease of the core inlet temperature. The phenomenon is qualitatively predicted by the TRACE model used. Fig. 12 shows the comparison between the experimental and calculated data for the difference of temperature between the fluid at the inlet of the core and the fluid at the exit of the SG primary side section.

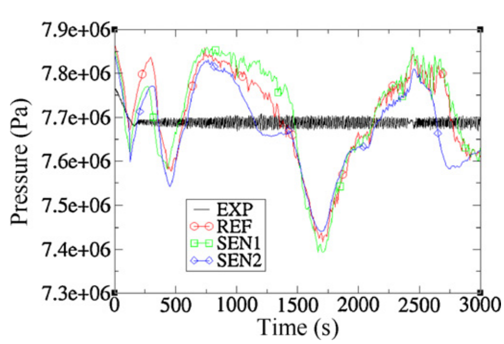

Fig. 11. Experimental data versus code calculations for PRZ pressure.

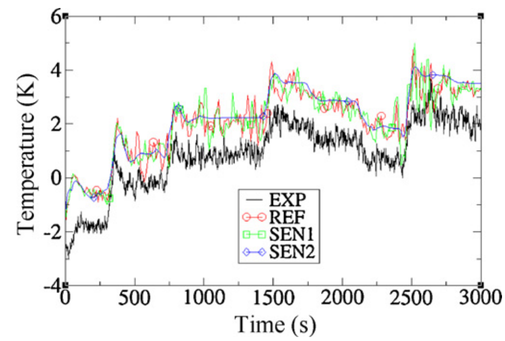

Fig. 12. Experimental data versus code calculations for the difference of fluid temperature at the inlet of the core and at the exit of the SG primary side.

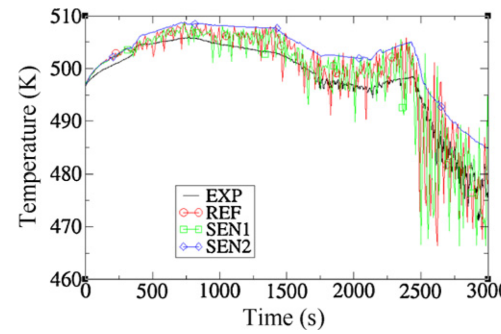

Fig. 13. Experimental data versus code calculations for the fluid temperature at the SG coil outlet.

The heat transfer in SG primary and secondary side and the superheating in secondary side phenomena are qualitatively predicted by the code. The average fluid temperature at the outlet of

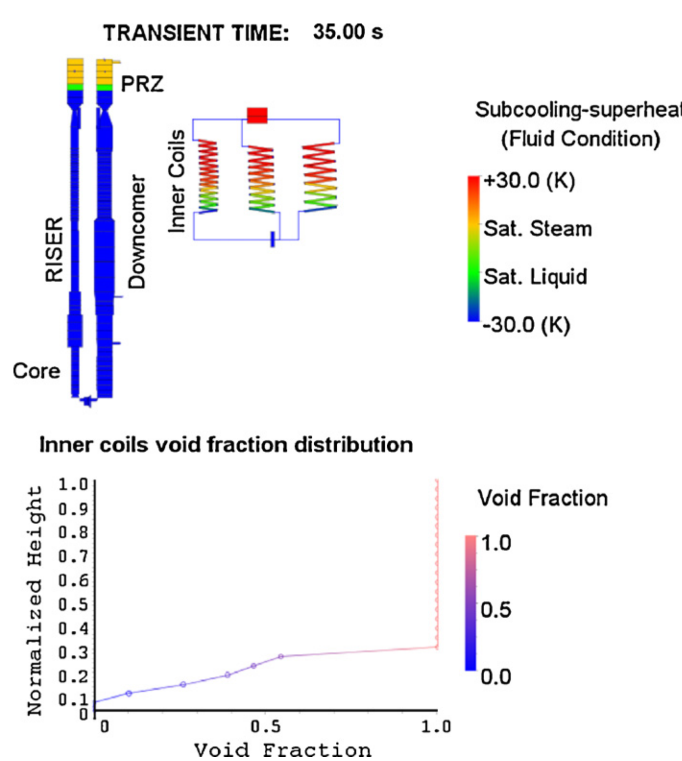

Fig. 14. SNAP animation model used to analyze the OSU-MASLWR-002 test.

the helical coils, Fig. 13, shows a qualitative agreement compared to the experimental data and a continuous oscillation though the secondary fluid is always in superheat condition. The temperature and pressure of the FW fluid, at the inlet of the SG, are imposed like boundary condition. The time dependence of the pressure, at the outlet of the SG, and the time dependence of the FW mass flow rate are imposed like boundary condition as well. A detailed model of the MS line (hydraulic volume, heat structures and related logic used to control the MS pressure) is necessary to further investigate the SG outlet pressure and temperature. The SNAP animation model, Fig. 14, shows the vertical distribution of the inner coils void fraction and the facility fluid condition during the transient. It shows that the secondary fluid enters subcooled at the bottom of the SG and boils off after travelling a certain length in the SG. In the TRACE model, in agreement with the experimental data, the steam will leave the SG superheated. As in the experimental data the slope of the MS superheat curve increases if the value of the core power increases and decreases if the value of the FW flow rate increases.

These analyses show that one of the reasons of the instability of the superheat condition of the fluid at the outlet of the SG, observed in a previous study (Mascari et al., 2008) as well, is the equivalent SG model used to simulate the different group of helical coils. In particular, if the helical coils are modelled by only one "equivalent" vertical tube (SEN2) a more stable fluid temperature at the outlet of the helical tubes is predicted by the code, Fig. 13.

4. Conclusion

The calculated data presented here are focused on a preliminary analysis performed by TRACE code, aiming at the evaluation of the code capability in predicting NC phenomena and heat exchange from primary to secondary side by helical SG in superheated condition and to evaluate the fidelity of various methods to model the SG in TRACE by simulating the OSU-MASLWR-002 test. Since the TRACE models are still not qualified, the current results are preliminary and should be not used for the code assessment, but are able to show the TRACE capability to reproduce the phenomena of interest in the selected test. Taking into account the test campaign already performed in the facility, the test OSU-MASLWR-002 gives the wider number of information about the single phase NC phenomena, the heat transfer in covered core, the heat transfer in SG primary and secondary side and the superheating in secondary side for a variety of primary and secondary operation conditions.

The analyses of the calculated data show that the phenomena of interest in the test are predicted by the code. The calculated results show a qualitative agreement with the experimental data for a number of main important parameters like inlet/outlet fluid core temperature, PRZ level, primary volumetric flow rate, delta T core and SG fluid outlet temperature. The PRZ pressure behavior shows discrepancies compared with the experimental data. A detailed analysis of the PRZ and steam line nodalization (hydraulic volume, active and not active heat structure, related logic used to control the PRZ and MS pressure respectively) is suggested in order to have a more accurate simulation of the selected test. The comparison between the experimental and calculated data shows that primary circuit stores more energy compared with the experimental data. One of the reasons could be an underestimation of the helical coil heat transfer coefficient during the different phase of the test. The analyses show that one of the reasons of the instability of the superheat condition of the fluid at the outlet of the SG is the equivalent SG model used to simulate the different group of helical coils. In particular, if the helical coils are modelled by only one "equivalent" vertical tube a more stable fluid temperature at the outlet of the helical tubes is predicted by the code. The model with three different oblique or vertical tubes needs more investigations, in order to study the possible instability conditions predicted by the code.

However in order to evaluate quantitatively the capability of the code in predicting NC phenomena and heat exchange from primary to secondary side by helical SG in superheat condition, and therefore to use these data for the TRACE code assessment, is necessary the qualification of several important operational characteristics of the facility including operational system/component heat losses and pressure drops.

References

Bonuccelli, M., D'Auria, F., Debrecin, N., Galassi, G.M., 1993. A methodology for the qualification of thermalhydraulic code nodalizations. In: Proc. of NURETH-6 Conference, Grenoble, France, October 5–8.

Boyack, B.E., Ward, L.W., 2000. Validation Test Matrix for the Consolidated TRAC (TRAC-M) Code. Los Alamos National Laboratory, LA-UR-00-778.

Bucalossi, A., 2006. Validation of Thermal-Hydraulic Codes for Boron Dilution Transients in the Context of the OECD/SETH and OECD/PKL Projects. EUROSAFE.

Cheng, Y.H., Wang, J.R., Lin, H.T., Shih, C., 2009. Benchmark calculations of pressurizer model for Maanshan nuclear power plant using TRACE code. Nuclear Engineering and Design 239 (2009), 2343–2348.

CSNI, 1996. CSNI integral test facility validation matrix for the assessment of thermal-hydraulic codes for LWR LOCA and transients, NEA\CSNI\R(96)17.

Freixa, J., Manera, A., 2010. Analysis of an RPV upper head SBLOCA at the ROSA facility using TRACE. Nuclear Engineering and Design 240 (2010), 1779–1788.

Galvin, M.R., 2007. OSU MASLWR Test Facility Modification Description Report. IAEA Contract Number USA-13386, Oregon State University, November 19, 2007.

Jasiulevicius, A., 2005. Benchmark Calculations of PKL Test E3.1 using TRACE code. PKL Analytical Workshop, Pisa, October 11.

Mascari, F., Woods, B.G., Adorni, M., 2008. Analysis, by TRACE code, of natural circulation phenomena in the MASLWR-OSU-002 test. In: Proc. Int. Conf. Nuclear Energy for New Europe 2008, Portoroz, Slovenia, September 8–11.

Mascari, F., Vella, G., Woods, B. G., Adorni, M., D'Auria, F., 2009. Analysis of the OSU-MASLWR natural circulation phenomena using TRACE code. IAEA, organized in cooperation with the OECD Nuclear Energy Agency and the European Commission, Technical meeting on "Application of Deterministic Best Estimate Safety Analysis", University of Pisa, Pisa, Italy, 21–25 September.

Modro, S.M., Fisher, J. E., Weaver, K. D., Reyes, J. N., Jr., Groome, J. T., Babka, P., Carlson, T. M., 2003. Multi-Application Small Light Water Reactor Final Report. DOE Nuclear Energy Research Initiative Final Report, Idaho National Engineering and Environmental Laboratory, December 2003.

Pottorf, J., Mascari, F., Woods, B.G., 2009. TRACE, RELAP5 Mod 3.3 and RELAP5-3D Code Comparison of OSU-MASLWR-001 Test. 2009 ANS Winter Meeting and Nuclear Technology Expo.

Queral, C., González, I., Expósito, A., 2008. Analysis of abnormal operation procedures in sequences of loss of the RHRS at midloop operation. Annals of Nuclear Energy 35 (2008), 1321–1334.

Reyes Jr., J.N., 2005. Governing equations in two-phase fluid natural circulation flows. Natural Circulation in Water Cooled Nuclear Power Plants, Annex 6, IAEA-TECDOC-1474, Vienna, Austria, November.

Reyes Jr., J.N., Groome, J., Woods, B.G., Young, E., Abel, K., Yao, Y., Yoo, J.Y., 2007. Testing of the multi application small light water reactor (MASLWR) passive safety systems. Nuclear Engineering and Design 237, 1999–2005.

Staudenmeier, J., 2004. TRACE Reactor System Analysis Code, MIT Presentation, Safety Margins and Systems Analysis Branch, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission (2004).

Symbolic Nuclear Analysis Package (SNAP), 2007. Users Manual. Applied Programming Technology, Inc., Bloomsburg, PA.

TRACE V5.0, 2008. Theory and User's Manuals. Division of System Analysis, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC.

Wang, J.R., Lin, H.T., Cheng, Y.H., Wang, W.C., Shih, C., 2009. TRACE modeling and its verification using Maanshan PWR start-up tests. Annals of Nuclear Energy 36 (2009), 527–536.

Woods, B.G., Mascari, F., 2009. Plan for an IAEA International Collaborative Standard Problem on Integral PWR Design Natural Circulation Flow Stability and Thermo-Hydraulic Coupling of Containment and Primary System During Accidents. Department of Nuclear Engineering and Radiation Health Physics, Oregon State University (Prepared for IAEA).